FINAL REPORT

PROJECT NO. A-530

DEVELOPMENT OF INSTRUMENTATION FOR BACKGROUND RADIATION MEASUREMENTS

By

WERNER N. GRUNE
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Research Assistant
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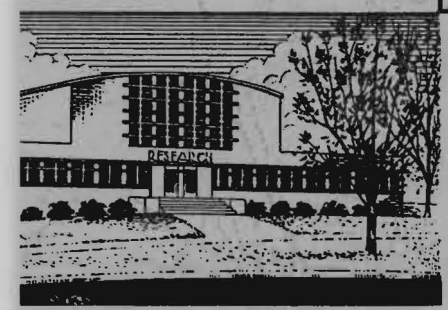
Covering the Period

26 October 1960 to 30 August 1962

CONTRACT NO. SAph 76183

Performed for the

Division of Radiological Health
UNITED STATES PUBLIC HEALTH SERVICE
Washington 25, D. C.



Engineering Experiment Station
Georgia Institute of Technology
Atlanta, Georgia

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ABSTRACT

A study of man's immersion dose of ionizing radiation, including gamma, beta, alpha and neutron radiation, was initiated. In conjunction with this study, monitoring equipment was developed for specific measurements of the background and fallout radiation components. Emphasis was placed on the three channel gamma energy analyzer. Radiation flux and air sample activity measurements were conducted utilizing the instrumentation developed under this contract and available sample analysis equipment.

In addition, theoretical and experimental studies were conducted to determine a method for the absolute calibration of a sodium iodide scintillator with gamma energies ranging from 0.1 to 1.0 Mev. The completion of these studies will provide a much needed calibration device.

The completion of the overall study program would provide an inexpensive and reliable gamma, beta, and neutron monitoring system capable of unattended operation at remote locations. In case of atomic attack, a network of these systems would allow precise mapping of the fallout pattern so that emergency measures could be implemented. In addition to this important objective, knowledge regarding the effects of chronic low-level exposure to ionizing radiation would be increased.

A program of fallout monitoring was conducted during the period October, 1961 to July, 1962, which also employed the gamma monitoring system designed under this contract.

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I. INTRODUCTION

In recent years much information has become available on the biological effects of radiation; maximum permissible concentrations in air, food and water; quantity and type of radiation emitted; the energy, half-life and size of nuclides; and their physical state. All this information has contributed toward the establishment of maximum permissible radiation exposures to man, but the effective biological dose still depends on many factors, some of them poorly defined.

As part of the problems to establish the biological effects of radiation, considerable speculation has evolved about the "threshold" exposure, namely, the level of ionizing radiation exposure below which no harm is caused. Theoretically any dose lower than "threshold" would be expected to cause no effect. However, the establishment of such behavior is dependent upon the sensitivity of the methods for detecting the effect. As more sensitive methods become available, the "threshold" dose might well be lowered, even approach zero. There is general agreement that genetic effects show a non-threshold behavior curve, i.e., there is no radiation dose below which genetic effects would not be observed. There is no agreement as to whether somatic effects are threshold or non-threshold in behavior (1).

As part of the problem assigned to the Federal Radiation Council to study biological effects of radiation, many efforts are underway to establish the best estimates of the harm resulting from given exposures to radiation. It is convenient to classify radiation effects either as high level (above 50 roentgens) or low level with special interest devoted to the region below about one roentgen. The evaluation of effects of

ionizing radiation at low levels, especially when the exposure extends over long periods of time, is much more difficult than evaluating high level radiation effects. There is no direct evidence on the effects of radiation under conditions of low-level long-term exposure.

The most useful approach to evaluation of low-level, long-term exposures, is to consider the background radiation to which mankind has always been exposed, then assume that additional exposure to double this dose will produce twice as much harm. However, it is important to recognize that the more our exposure increases over background, the more uncertain we are about the effects. Essentially this means that if any amount of radiation produces harm, then the radiation from our natural environment, namely background radiation, must have been producing harm to mankind through the centuries. Available information on background radiation can therfore perhaps serve as the best yardstick.

Since the beginning of time, man has been exposed to external radiation from cosmic rays and from naturally-occurring radioactive materials such as uranium, thorium and potassium-40 in the ground, air and structures in which he lives and works. Man has also been exposed to internal radiation from radionuclides that occur universally in the body, for example, potassium-40, carbon-14 and members of the radium and thorium families. It is estimated that the typical radiation dosage from such natural sources is about one-tenth of a roentgen per year, amounting to about seven roentgens over a standard 70-year lifetime (2). It must be pointed out that, in some areas of the world, in parts of India and Brazil, for example, the natural radiation exposure may be up to ten timeshigher. Population studies are being conducted to see if any effects can be detected. Such studies are most

difficult, however, because one needs to have controlled populations for comparison; populations that have the same standard and way of living, the same nutritional plane, the same medical care, the same degree of inbreeding. As yet, in these areas there is no evidence that the backgrounds ten times higher than average are deleterious, but this does not mean that there are no affected individuals. It does mean that the number of individuals affected may be too small to be detected.

A report of the Subcommittee on Reorganization and International Organizations of the United States Senate, entitled "Radiation Research in the Life Sciences" (3), represents the activities on world-wide radiation research. The purpose of the compendium was to explore the beneficial and harmful effects of radiation on living systems and on molecular systems important to life.

A background survey was made over a large part of the United States by Solon, et al. (4). This transcontinental survey, conducted by personnel assigned to the New York Operations Office, United States Atomic Energy Commission, found the external environmental radiation dose rates in populated areas of the United States to range from 70 to 175 mrad/year. The measurements taken with a large ionization chamber were used to determine the radiation levels at different geographical and topological locations throughout the country. The results of this survey, although not highly quantitative, offer useful information.

Through cooperation with the National Nuclear Energy Commission of Brazil, it is recognized that there are two geographical areas in Brazil with relatively high background radiation intensities, one monazite, which averages a dose of 0.5 rad/year to about 50,000 inhabitants, and one a

mineralized volcano area which averages about 1.6 rad/year to a sparse population. With the thought that such areas of pupulations might provide material for studying the effects of chronic low-level exposures, a Commission of Human Genetics has been created.

Because the most urgently needed data are those obtained from man himself, the World Health Organization has been assisting with the planning of studies on large population groups living in areas of high natural radiation (5). The W.H.O. report has placed special emphasis on the Karela and Madras monazite regions of India, where the background radiation dose measures approximately 1.3 to 1.5 rad/yr for about 80,000 people. These populations thus receive 10 to 15 times as much ionizing radiation as most people and are estimated to be within the range necessary to double the mutation rate.

In this connection, Weber (6) has stated that the radiological health program is moving deeper into the operational phase, as indicated by consistent downward revision over the past thirty years of the annual maximum levels of ionizing radiation considered permissible by the National Committee on Radiation Protection and other authoritative groups. It has dropped from 60 rems for workers occupationally exposed (in 1931-36) to 30 rems (1936-48) to 15 rems (1948-58) to 5 rems at the present time. Each downward revision has increased the responsibility of those concerned with the problems of protection from radiation.

In conclusion, many studies have been conducted to measure a particular component of the background radiation, but there is no evidence of a comprehensive study in which a number of the components of background radiation were considered simultaneously.

II. STATEMENT AND PURPOSE OF THE RESEARCH

A. The Problem

The study of background radiation has received considerable attention, but a comprehensive investigation of all aspects, including both natural and man-made sources of radiation at ground level is lacking. A study of these sources is necessary to accurately determine man's immersion dose of ionizing radiation in the environment. Such studies may establish significant ratios between the several components of what may be loosely termed "background".

In order to solve this problem adequately, an instrumentation system of sufficient accuracy to measure the several components of the background must be developed. An important requirement is the necessity of developing a reliable and inexpensive system, so that it can be produced in sufficient quantity to make measurement of the radiation on a world-wide basis economically feasible.

B. Purpose of the Study

The purpose of this study is to determine quantitatively the four major components of the environmental immersion dose, viz., gamma, beta, neutron and alpha radiation. It is anticipated that a better understanding of background radiation will result from these studies and valuable techniques for the quantitative measurement of background radiation will be developed.

C. Background

Ionizing radiation from outer space, in the form of cosmic rays, was present long before life itself began on earth. However, spontaneous emission of energy by matter was not discovered by man until 1896. This

research project has been concerned with investigating the ionizing radiation produced by the above two processes. Two major divisions of the background radiation have been logically made at the outset: cosmic and terrestrial radiation.

1. Cosmic Radiation

The term, cosmic radiation, is applied to all radiation of cosmic origin. Every second nearly 10^{18} cosmic-ray particles plunge into the earth's atmosphere from space. The overwhelming majority have an energy of about 5×10^{9} electron volts. A very small fraction have energies of the order of 10^{18} electron volts. On the average, a primary cosmic ray travels through only about a tenth of the atmospheric blanket before it hits an atomic nucleus. The collision produces a number of secondary particles; some are fragments of the struck nucleus and others are created at the moment of impact. Many of these "secondaries" strike other nuclei in the atmosphere, giving rise to still more particles, and so on. Thus, a single primary ray initiates a "shower" of many particles.

At each successive generation the energy of the particles decreases. If the primary particles had energy on the order of 10^{10} electron volts, the shower will die out before it reaches the ground. The predominant particles from these showers are electrons, while mu mesons are the second most abundant. As they descend, the atmosphere scatters them. By the time the shower reaches the ground it is spread over a wide area. The density of particles is greatest near the center, the point where the primary would have hit if there had been no collision. However, even at large distances from the center of the shower, there are enough particles to measure. A detector with a sensitive area of one square meter will respond to a shower of 10^{6} particles out to about 150 meters from the center, and to a 10^{8} particle

shower out to about 700 meters (7). There are several variables affecting cosmic radiation:

a. Altitude - a dependence of cosmic radiation on altitude is shown in Figure 1.

b. Temperature - meson activity increases with increasing temperature at the rate of 0.12% per ^{o}C.

c. Moon phase.

d. Solar disturbances.

e. Latitude[1].

f. Barometric pressure[2].

2. Terrestrial Radiation

Until the work of the Joliet-Curies in 1932, who produced the first aritficial radioisotopes, the natural radioisotopes stood alone (10). In this study only naturally occurring radioactive materials and fallout have been considered as the major contributors to terrestrial radiation.

Of the naturally occurring radioisotopes, only four (U-238, Th-232, and their daughter products, K-40 and C-14) constitute the overwhelming bulk of natural radioactivity in the environment. The distribution of these isotopes within the earth depends on their chemistry rather than on their

---

1 The dependence of cosmic radiation on latitude was discovered by Clay (8) in 1927. This dependence, shown in Figure 1, is due to the variation of the earth's magnetic field with latitude. There is also a slight variation with longitude.

2 Over the normal range of barometric pressure, the intensity of cosmic radiation varies inversely. Attenuations result directly from the corresponding air density changes. The constant of proportionality is about one or two per cent per centimeter of mercury (9).

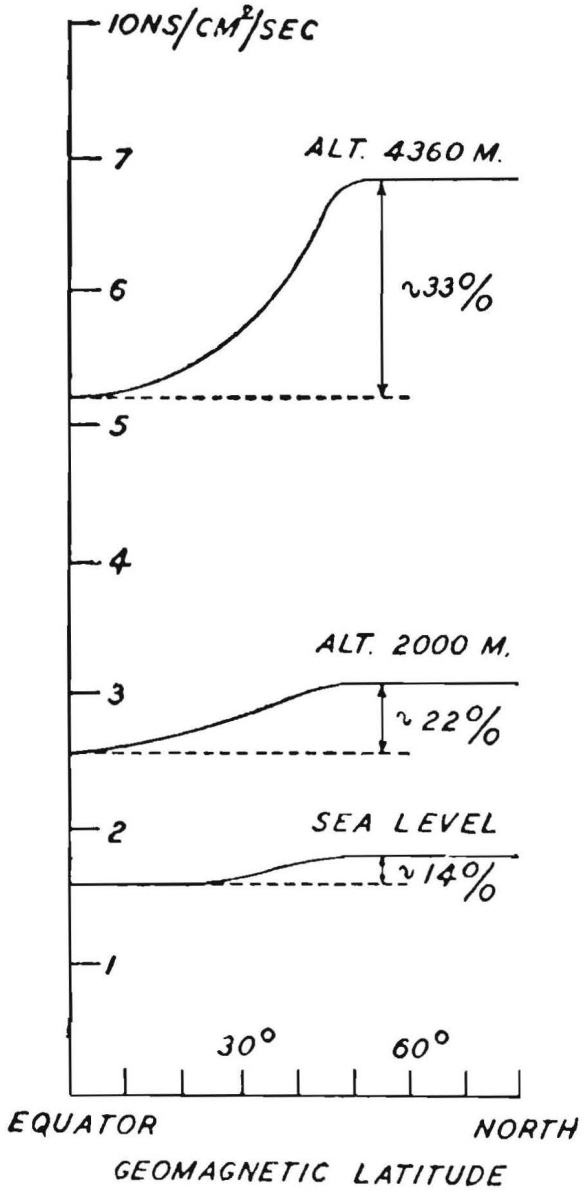

Figure 1. The Effect of Altitude and Latitude on Cosmic Radiation Intensity.

nuclear properties. Their oxides are relatively low in density and thus occur in the crust of the earth, rather than in its dense mantle and metallic core. It is estimated that one-half of the radioactivity in the earth lies within forty miles of the surface. This concentration significantly increases environmental radioactivity above that due to cosmic radiation.

The radiation from isotopes in the crust is rapidly attenuated as it passes through rock and soil. Most of the activity that makes up the environmental background originates less than six inces below the surface. The average surface concentration is on the order of two curies per square mile, though values vary by a factor as large as five in different locations (10). The contribution to surface activity from nuclear testing has increased this activity by another 100 millicuries of Cs^{137} and 50 millicuries of Sr^{90} per square mile (11). In general, the more acid rocks such as pegmatites (12) and granites are more radioactive than the alkaline basalts, and limestones which generally show lower radiation levels (13)(14).

Through decay, radon is produced and diffuses out of the soil into the atmosphere. For the purpose of this project, the variables affecting the diffusion rate are:

a. Height

b. Meteorology

c. Location (city, country)

d. Geology.

Figure 2 depicts the components of environmental radiation. Figure 3 constitutes a small segment of Figure 2. It illustrates the interaction of cosmic rays with the atmosphere.

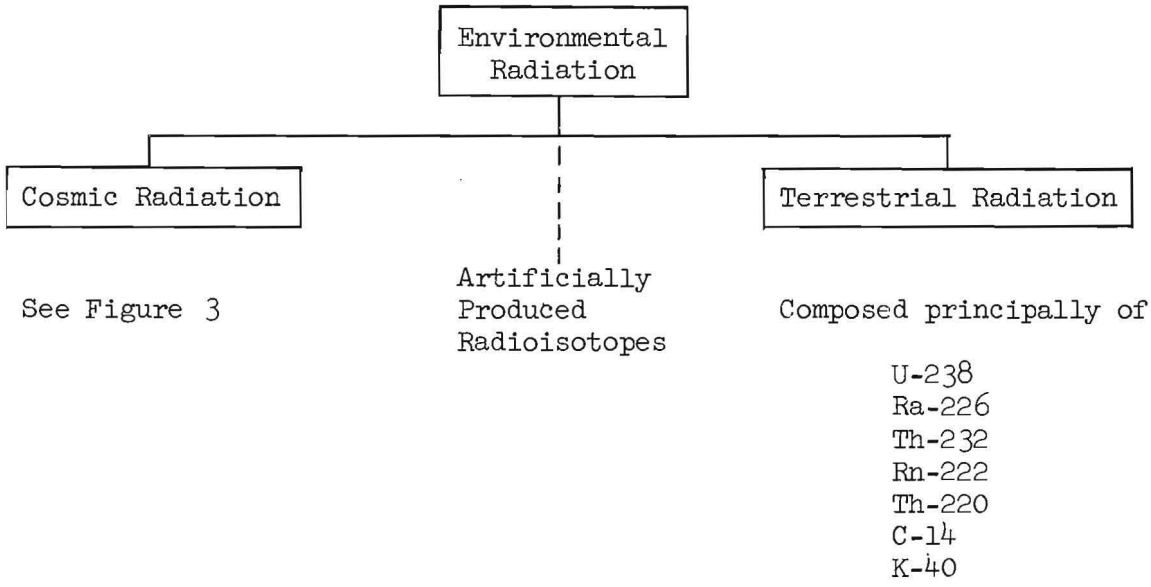

Figure 2. Components of Environmental Radiation

Figure 3 shows the origin and the mechanisms by which the cosmic radiation is produced. It may be noted that a primary particle, usually a proton, collides with an oxygen or nitrogen nucleus in the atmosphere. This event can result in the products shown on the chart and is called a shower development. These products, if not stable, decay according to the schemes and mean lifetimes indicated. The end products of interest were gamma rays, electrons and neutrons.

The mechanism of neutron production from cosmic radiation is of special interest. Their production, as well as their disappearance, is indicated in the lower left-hand corner of Figure 3.

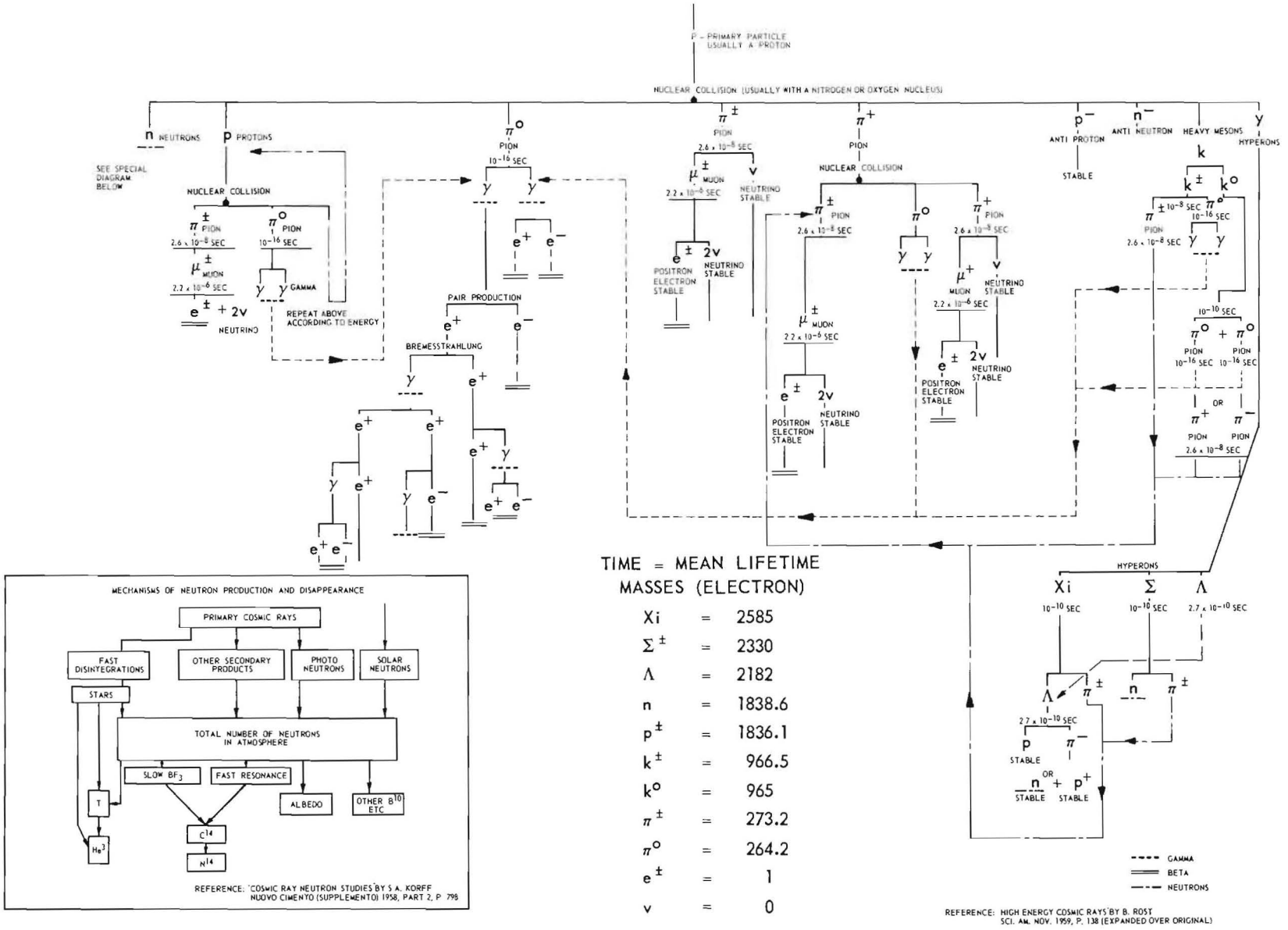

Figure 3. Interaction of Cosmic Radiation with the Atmosphere.

III. RESEARCH AND DEVELOPMENT WORK

A. The Gamma Detection System

To measure man's immersion dose of gamma radiation, a detection system employing a sodium iodide scintillation crystal, a three channel pulse height analyzer, and the associated instrumentation for continuous monitoring have been developed under this contract (15)(16)(17). The system divides the observed scintillations into three light intensity ranges. These correspond to the light intensities produced in the crystal by electrons undergoing the photoelectric effect with a gamma photon. The three intensity thresholds in the present instrument correspond to energies of photoelectrons ranging from 0.1 to 0.5 Mev, 0.5 to 1.0 Mev, and 1.0 to ∞ Mev.

Schematic and block diagrams of the complete gamma analysis unit are shown in Figures 4, 5, 6 and 7.

Although plastic scintillators are cheaper than sodium iodide, their photoelectric cross section is too low in the gamma energy range under study to be of much value for energy analysis. Since absorption by the Compton effect is the predominant gamma interaction phenomenon in plastic scintillators, the apparent energy of a given gamma photon will vary over a wide range.

1. Design and Operation of the Gamma Detection System

The three channel pulse amplitude analyzer and gamma ray detection system developed at Georgia Tech consists of four major parts. These are (i) the detection unit, (ii) the three channel pulse height analyzer, (iii) the count rate circuit and recorder multiplex unit, and (iv) the power supply. A block diagram of the system is shown in Figure 4.

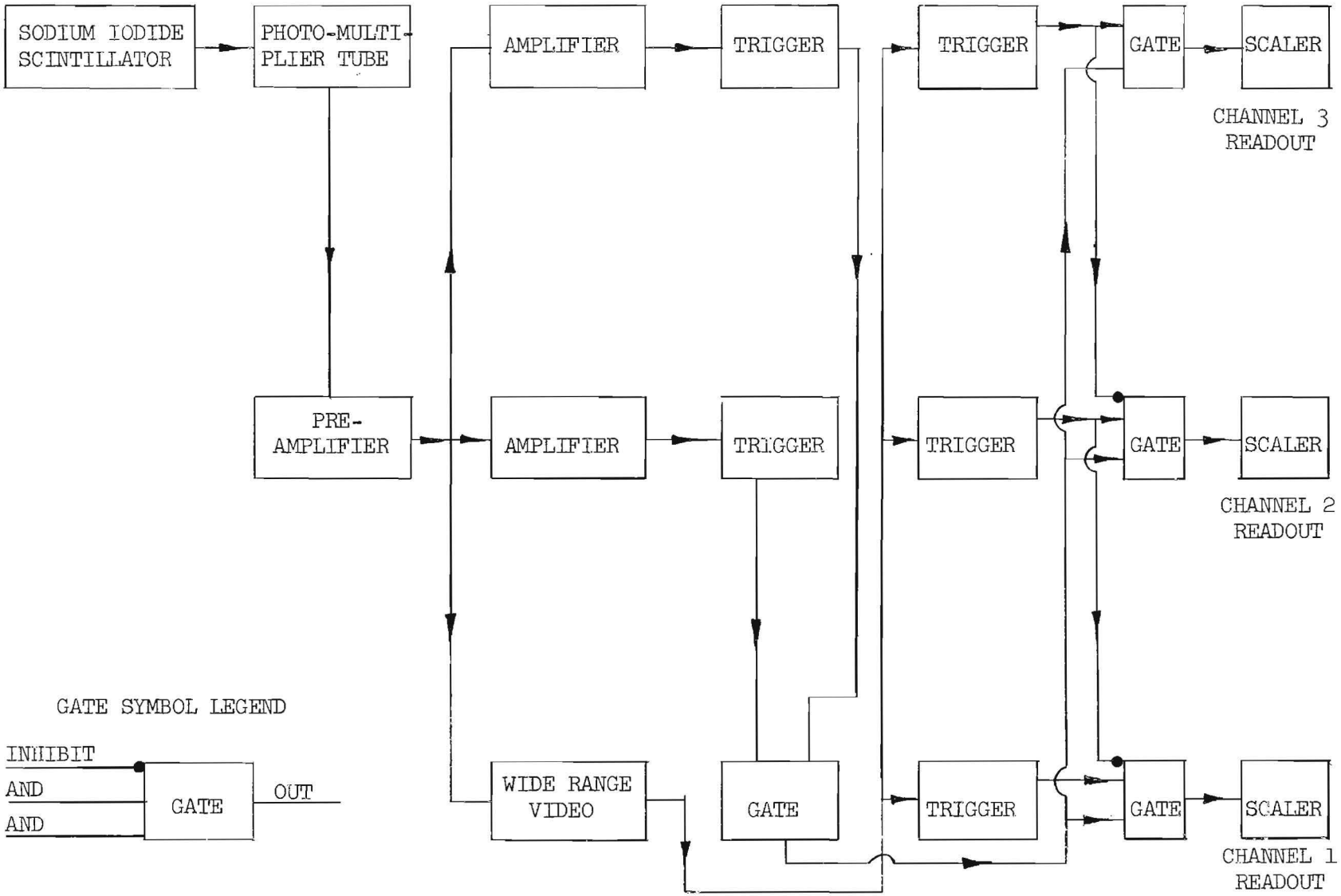

Figure 4. Block Diagram of Gamma Detection System

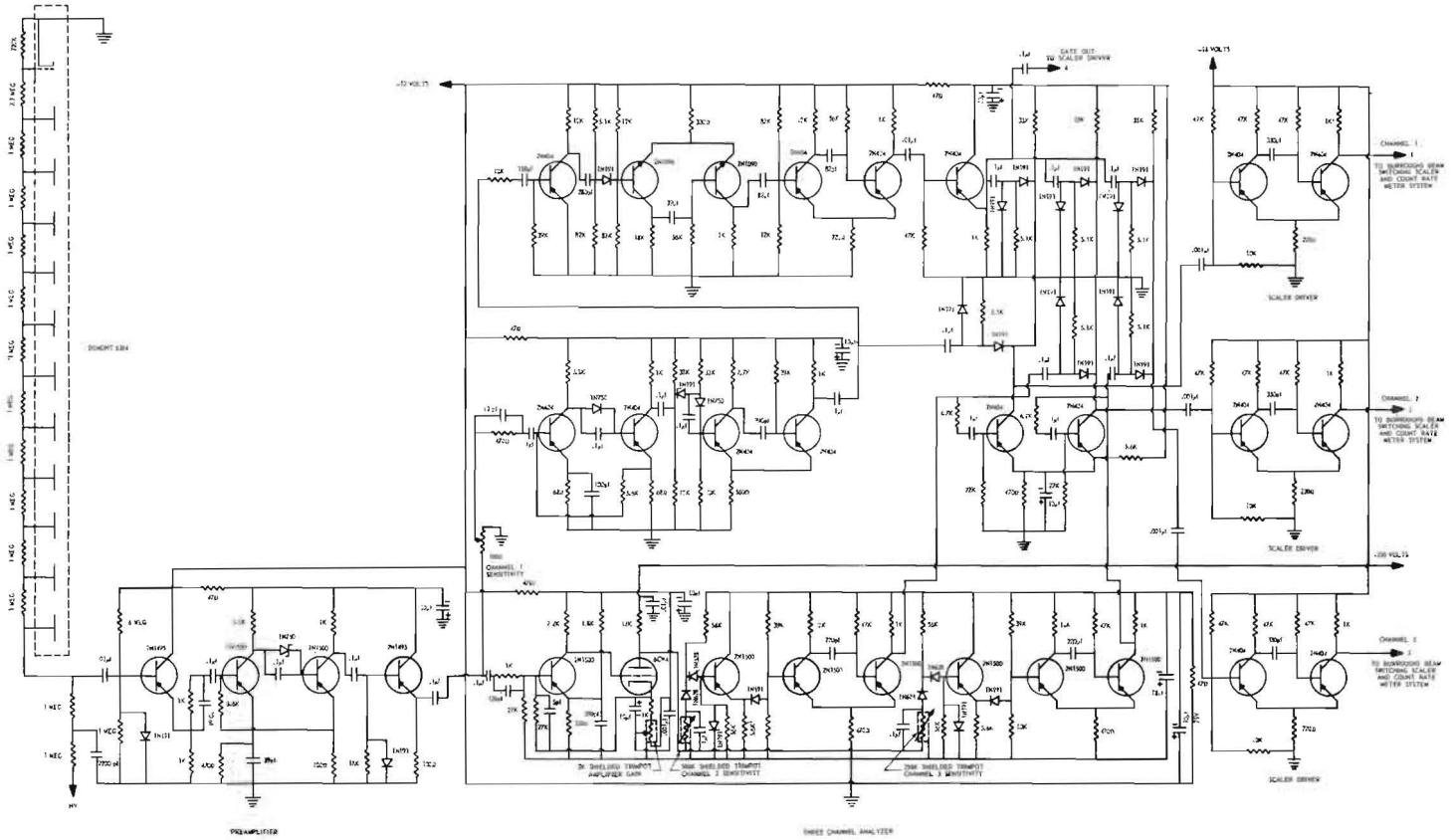

Figure 5. Schematic Diagram of Amplifier and Logic Sections of Three Channel Gamma Analyzer.

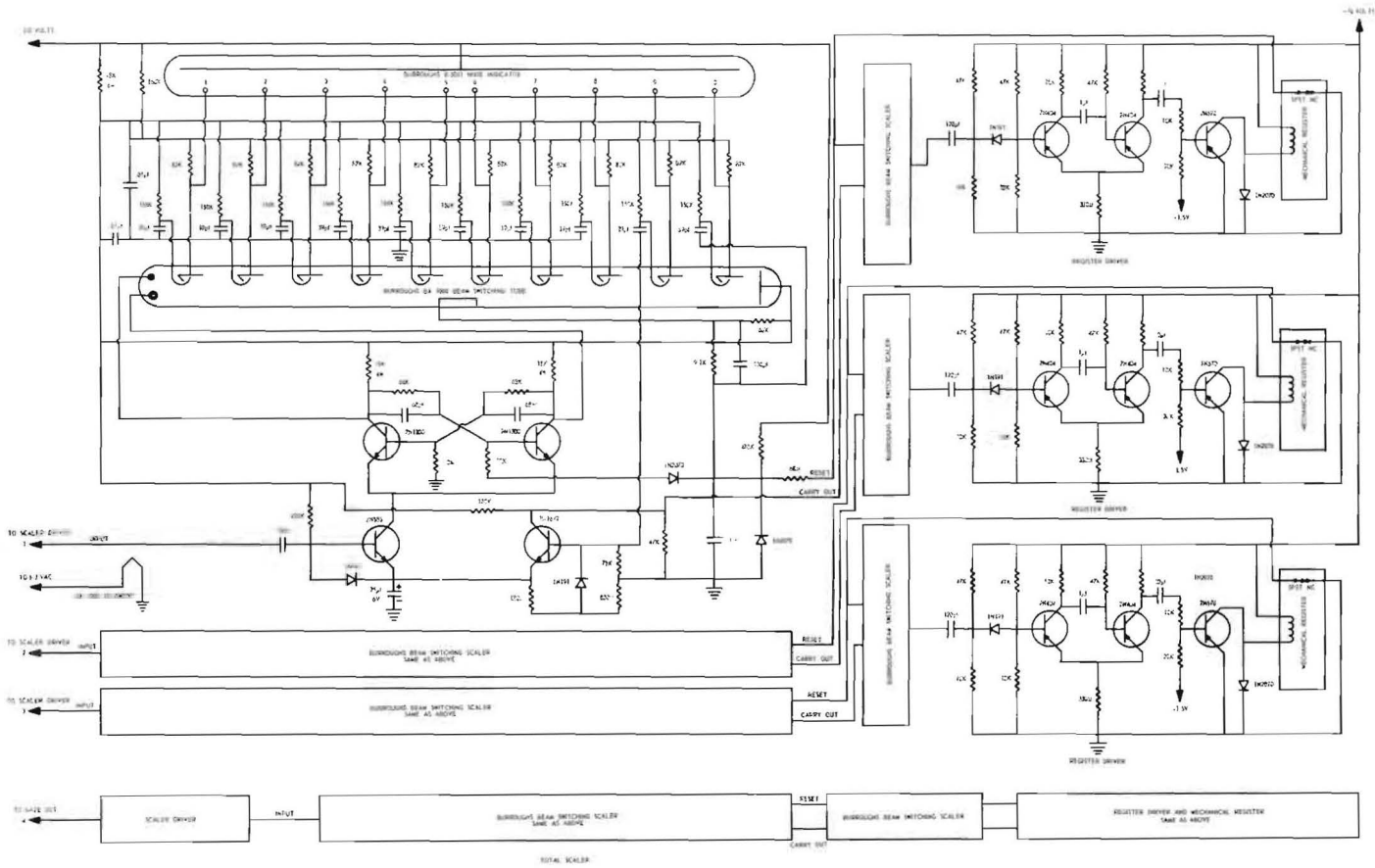

Figure 6. Schematic Diagram of Scaler Portion of Three Channel Analyzer.

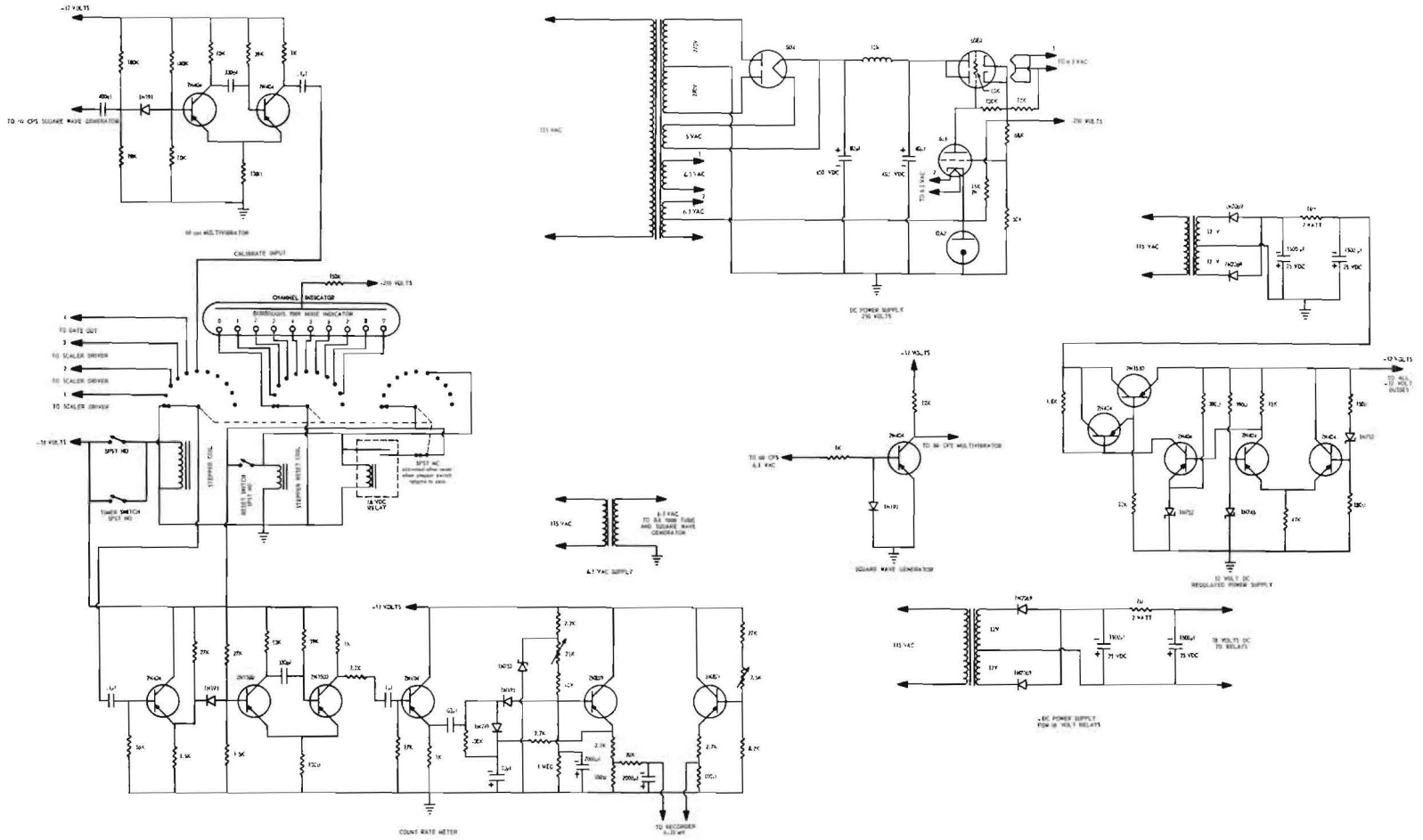

Figure 7. Power Supply and Count Rate Meter Section of Gamma Analyzer.

(a) The Detection Unit consists of a two inch photomultiplier tube[3] optically connected to a Harshaw Type 7F8 two inch scintillation crystal[4].

These components, as shown in Figure 8, along with the necessary transistor preamplifier were assembled in a light-tight box for installation in any desired location. The light-tight box was constructed of plywood and surrounded with a thin copper foil to reduce electrical interference.

The detection unit was connected to the other parts of the gamma analysis system by 93 ohm coaxial cables and twisted pairs. An electrical interlock prevented opening the light-proof box with the photomultiplier high voltage on.

The detection unit was temperature controlled and the electronic circuits were temperature compensated to eliminate any errors in calibration from this source. No observable change (less than 1%) occurred for temperature variations between 0°C and 75°C, which included all normal ambient temperature ranges. The schematic diagram in Figure 5 shows the preamplifier circuit for this unit.

(b) The Three Channel Pulse Amplitude Analyzer accepts pulses from the photomultiplier preamplifier and quantizes these pulses into three energy bands. A block diagram of the three channel analyzer, as used in the fallout monitoring program here at Georgia Tech, is shown in Figure 4.

The theory of operation is as follows. A signal from the photomultiplier preamplifier is fed to the input of the analyzer. The two input amplifiers have been paralleled so that only one input is needed instead of the two as in the original circuit (15).

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[3] Dumont type 6292, manufactured by the Allen B. Dumont Laboratories, Inc., 750 Bloomfield Avenue, Clifton, New Jersey.

[4] Product of Harshaw Chemical Co., 1945 East 97th St., Cleveland 6, Ohio.

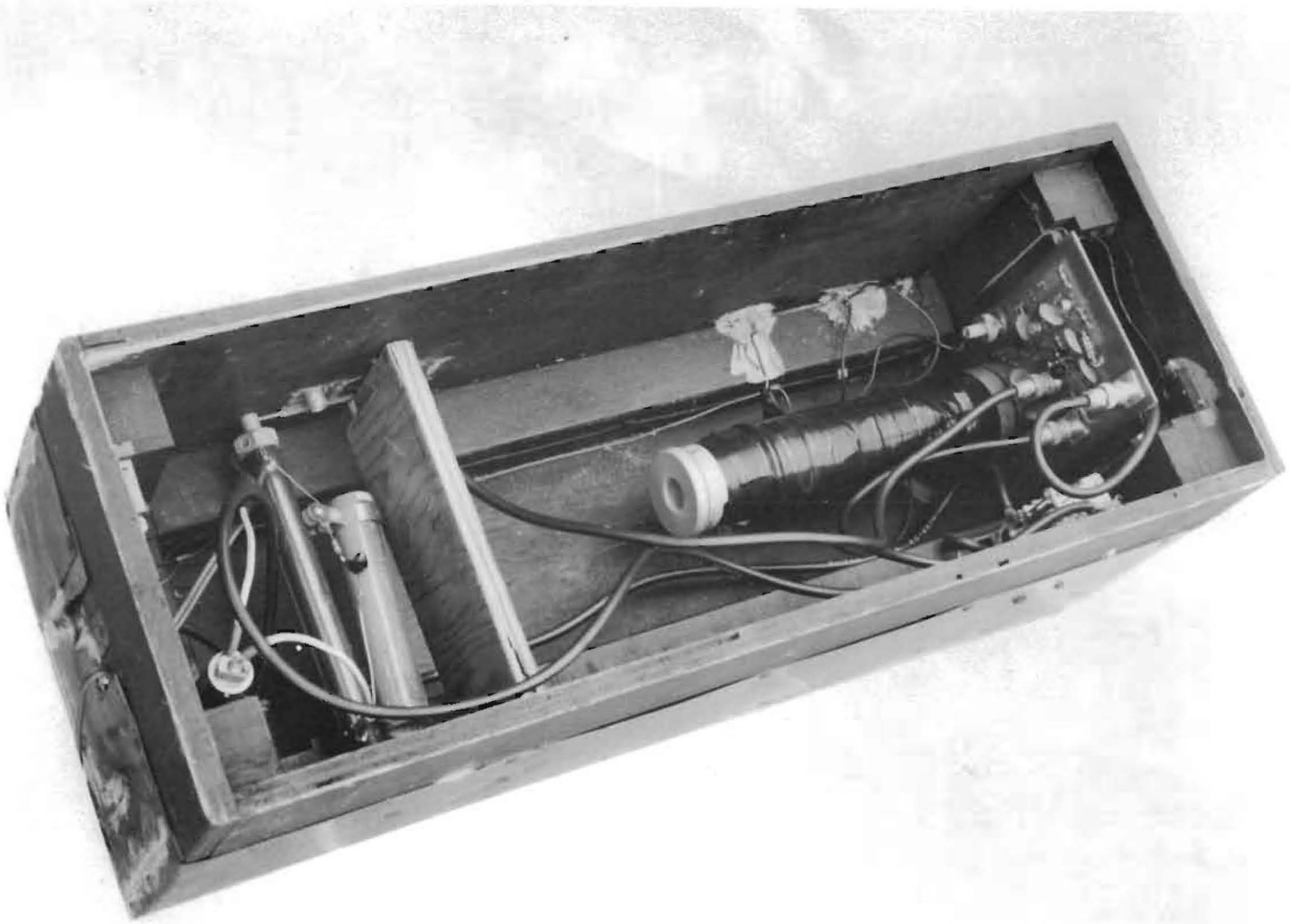

Figure 8. Gamma Detection Unit.

The amplifiers feed three discriminator circuits each of which is set at a trigger level corresponding to some gamma energy. The input circuitry to these discriminators utilizes a unique diode arrangement to eliminate the problem of loading of the amplifier after a trigger circuit has been switched on. The outputs of these discriminator circuits go to a gating network where the correct energy range is selected. This range selection is accomplished by each gate being turned on by its associated trigger and off by the trigger having an energy threshold at the next higher level.

The outputs from these gates are then fed to scalers which totalize each energy band. In addition, an overall total is obtained by feeding the output from the channel one discriminator to a fourth scaler.

(c) The Count Rate Integrator and associated strip chart recorder[5] along with the sequencing relay and stepping timer, as shown in Figure 7, provide time multiplexed readout. This section provides a full cycle including the sampling of channels 1, 2, 3, total and two calibration checks during each 30 minute period. The output is recorded on a strip chart recorder for subsequent analysis.

(d) The Regulated Power Supplies, shown in Figure 7, provide the necessary power to the electronic circuits and multiplexing unit. A photograph of the entire gamma analysis unit is shown in Figure 9.

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5 Model 8000-2200, Wheelco Recorder, a product of Barber-Colman, Co., Rockford, Illinois.

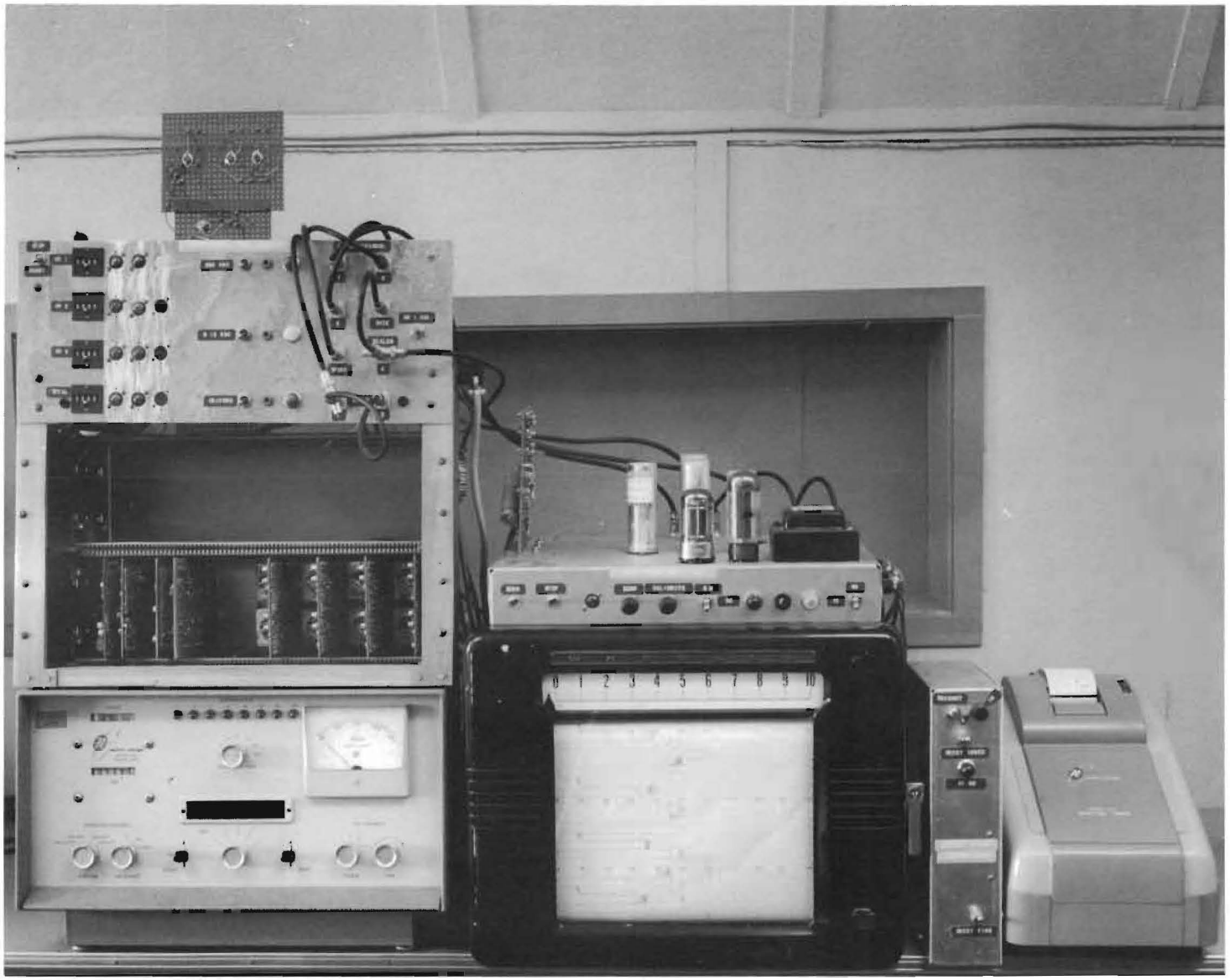


Figure 9. Complete Gamma Analyzer System.

B. Calibration of Scintillation Detector

During a study of the problem of calibrating the sodium iodide scintillator to measure the energy of incident gamma radiation, several difficulties were discovered. These difficulties may be explained by examining the three processes of interaction involved in the conversion of gamma ray energy to light energy (see Figure 10). The three processes involved are discussed below:

(a) In the photoelectric effect, the incident gamma ray interacts with an atom and all of its energy is transferred to an electron. The electron is ejected from its energy state with a kinetic energy given by:

$$K.E. = h\nu - E_b \qquad (1)$$

where $h\nu$ is the total energy of the incident photon and E_b is the binding energy of the electron. The electron which is usually ejected is the one from the K-shell. The electron thus ejected will wander about the crystal lattice until it encounters an imperfection in the crystal. When it encounters such a defect, it may drop into an unfilled shell of the associated atom, giving off light energy in the process. The electron may also expend its energy in some other non-nuclear process, possibly in the form of thermal energy. The binding energy of the electron in the original atom is small with respect to the energy, $h\nu$. Therefore, for the photoelectric effect, a good output light energy versus input gamma energy exists for gamma rays having $h\nu \gg E_b$. In practice, photoelectric absorption of gamma radiation is important only for energies less than 1 Mev, and more for absorbers of high atomic number. An approximation to the probability of interaction for the photoelectric effect is given below.

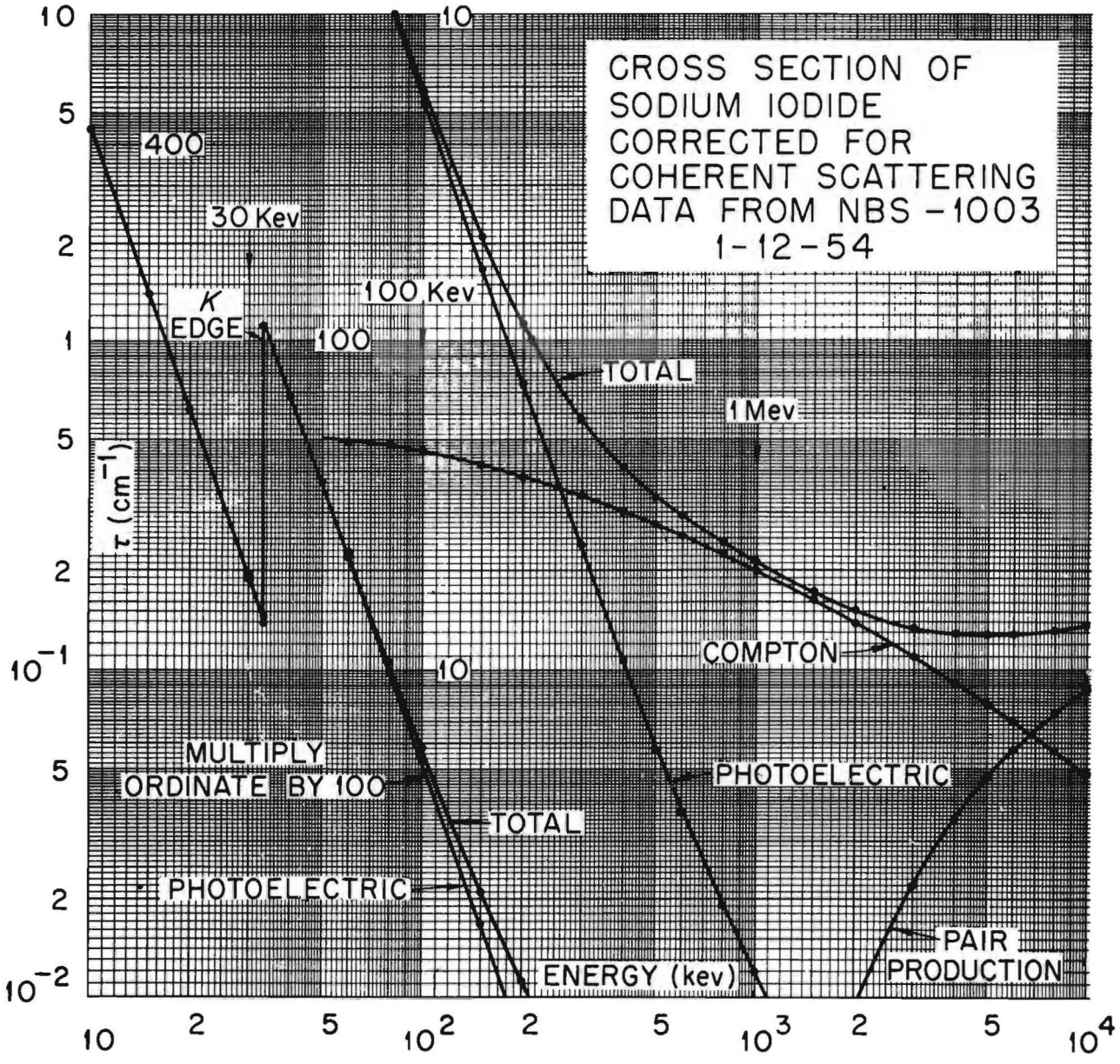


Figure 10. Cross Section of Sodium Iodide Corrected for Coherent Scattering.

$$P = k \times \frac{Z^n}{E^3} \qquad (2)$$

P = probability of occurrence

n $\doteq$ 3 for low energy gamma rays

n $\doteq$ 5 for high energy gamma rays

Z = atomic number.

(b) In the Compton effect, the primary photon (gamma ray) may interact with any electron, but not all the energy is transferred to the electron. The most probable interaction is with the valence, or very loosely bound, electrons. Essentially, the interaction may be considered as an elastic collision between the primary photon and the electron. The result of this collision is that the energy and momentum of the primary photon is shared between a secondary photon of energy $h\nu'$ and a recoil electron. An approximation of the Klein-Nishina formula (18) for the probability of Compton interaction is given below, although a much more rigorous treatment of the Compton effect is presented later.

$$P \doteq k \times \frac{Z}{E} \qquad (3)$$

This division may occur from zero energy transferred to the electron up to some maximum energy less than the energy of the incident photon. (When the maximum energy is transferred to the electron, the scattered photon recoils in the opposite direction to the travel of the recoil electron). The recoil electron then falls into a lattice defect in the same manner as does a photoelectron, emitting light in the process. In the Compton process, however, the energy of the incident photon has no direct correspondence to the energy of the recoil electron. Thus, no quantitative

energy analysis of random energy gamma photons is possible using conventional techniques in the energy range where the Compton process is significant.

(c) The third process of interest is pair production. In this process a gamma ray possessing energy greater than $E_{min} = 2\ m_o c^2$ may cause a positron and an electron to be produced in the coulomb field of a nucleus. The total kinetic energy of the pair will be:

$$K.\ E. = h\nu - 2\ m_o c^2 \tag{4}$$

$$2\ m_o c^2 \doteq 1.02\ \text{Mev} \tag{5}$$

Therefore, pair production cannot occur for a gamma quantum having energy less than $\doteq$ 1.02 Mev. The electron resulting from this process will have a kinetic energy slightly less than the positron since the positron is repelled by the coulomb field of the associated nucleus, while the electron is attracted. The electron will have energy approximately given by (but slightly less than) $E_{electron} = \frac{1}{2}(h\nu - 2\ m_o c^2)$. As in the case of the other processes, the electron may fall into a lattice defect causing fluoresence of the scintillator. After the positron loses its kinetic energy to the lattice, it becomes annihilated with an electron. This interaction generates two (or possibly more) photons which travel in opposite directions (in the case where two photons are generated), each having $K.\ E. = m_o c^2$, or $K.\ E. \doteq 0.51$ Mev. These photons may again interact with the scintillator. Pair production, like the Compton effect, does not release the full energy of the incident photon (gamma ray) and thus this process is not suitable as a direct measurement of the energy of the incident photon. As may be seen from equation (6) the probability of pair production increases with the atomic number of the absorber.

$$P \doteq k \times Z^2 (E - 1.02) \qquad (6)$$

The problem of calibrating a scintillation detector as an energy discriminator becomes obvious when one considers that in the range of interest (0.1 Mev and up) the three processes overlap.

As an example, consider a 0.5 Mev gamma incident on the scintillator. According to Figure 10, the probability for a photoelectric interaction as compared with the total probability of interaction is P = 0.18 and the probability for Compton interaction as compared to the total probability of interaction is P = 0.82. Obviously, the Compton process predominates and yet it is only possible to obtain an accurate indication of photon energy from the photoelectrons. The electrons accelerated by the Compton effect may have kinetic energy from zero up to some maximum less than the 0.5 Mev incident gamma. However, the scintillation detector is useful for isotope identification because a standard sample can be used for calibration and one can observe a photoelectric peak with an analyzer. With this known spectrum an unknown sample can be evaluated by comparison.

However, there is no such "standard" with which to compare a background spectrum. The spectrum obtained on single or multi-channel analyzers has a photopeak which corresponds directly to the energy of the photons emitted from the samples. The remainder of the "spectrum" presented on the output comes from Compton shifted photons and/or pair production. Of course, these last two conversions are not useful for direct energy analysis.

A thorough study has been made to determine a method of overcoming this problem and the proposed method is described in the following section.

C. Compton Shift Eliminator Study

It has been shown that it is not feasible to determine the energy distribution of background gamma flux by the use of a single detector. There is no technique with which to determine whether a pulse of a given height resulted from a photon producing a photoelectric effect, in which case the pulse height would be proportional to the gamma ray energy, or from a photon which had some undetermined higher energy undergoing Compton scattering or pair production. Since the last two events predominate over the photoelectric effect over most of the gamma energy band, the pulse height information from a NaI crystal is almost useless for determining the actual energy distributions of cosmic radiation.

To overcome these difficulties, a system is proposed which will disregard most of the pulses resulting from gamma reactions other than the photoelectric effect. The reaction of main concern is the Compton effect. In the Compton effect, part of the gamma ray energy is absorbed by an electron in the NaI crystal and the remainder may escape the crystal in the form of a scattered photon. It may be assumed that the maximum scattering angle is not much greater than 90 degrees and that the probability for scattering at greater than 90 degrees is low. Under these assumptions, photons which do scatter at greater than 90 degrees will have lost so much energy in the process that they would probably undergo a photoelectric reaction before escaping the crystal. Thus, the desired result would be obtained anyway.

In the proposed scheme, a shielding scintillator would surround the NaI and almost all of the scattered photons which escape the NaI would enter the shield surrounding the sides and bottom of the crystal.

This shield would be of sufficient thickness to insure a very good probability of some sort of light producing interaction with the scattered gamma rays. By observing both the shield and the NaI with separate photomultiplier tubes and rejecting all time coincident pulses from the two tubes, one effectively rejects most of Compton pulses from the NaI detector. This interaction can also take place in reverse, i.e., the scattered photon going from the shield into the NaI. However, these pulses are still coincident and therefore will be rejected. This system would eliminate many of the pair production pulses, although laboratory studies have shown that there are not many background photons of sufficient energy for pair production. The pulses remaining after coincidence elimination would consist principally of photoelectric pulses and may be pulse height analyzed for gamma ray energy.

The overall counting rate would be reduced, but most of the counts lost would have produced Compton shifts in the NaI scintillator and therefore would have been useless from an energy measurement standpoint.

A theoretical study has been made utilizing the Klein-Nishina formula (18) for the differential cross section per electron.

The differential cross section per electron as given by Davisson (19) and Siegbahn (20) and Nelms (21) is:

$$\frac{d\psi}{d\Omega} = \frac{r_o^2}{2} \left(\frac{\epsilon'}{\epsilon}\right)^2 \left(\frac{\epsilon}{\epsilon'} + \frac{\epsilon'}{\epsilon} - \sin^2\theta\right) \qquad (7)$$

where

ψ is the Compton cross section,

Ω represents solid angle,

θ is the angle of the scattered photon,

$r_o = \frac{e^2}{m_o c^2}$, the classical radius of the electron,

$\epsilon' = \epsilon/[1 + \epsilon(1 - \cos\theta)]$,

$\epsilon = h\nu/m_o c^2$,

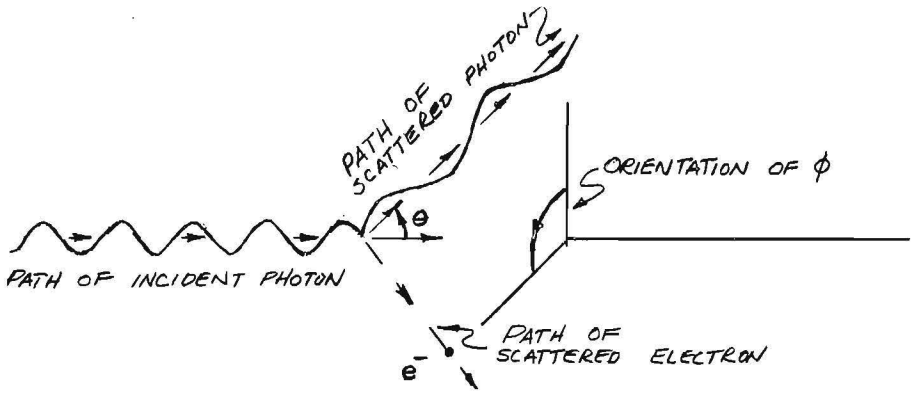
$h\nu$ is the energy of the incident photon (gamma ray),

$m_o c^2$ is the rest energy of the electron.

An analytic integration of the above equation was made to check the results obtained from the references (19)(20)(21) which did not agree. The correct results were found in the papers by Davisson (19) and Nelms (21). The differential cross section integrated over all values of θ and ϕ, i.e., over 4π steradians, is:

$$\Psi = \frac{\pi r_o^2}{\epsilon^2} \left[\left(\epsilon - 2 - \frac{2}{\epsilon} \right) \ln (1 + 2\epsilon) + \frac{2(\epsilon^2(\epsilon+1)+2(1+2\epsilon)^2)}{(1 + 2\epsilon)^2} \right] \quad (8)$$

Equation (7) was integrated over different sets of limits to determine the per cent of the photons that would be scattered over the various values of θ. Figure 11 illustrates the geometry.



Geometry of Compton Interaction
Figure 11

A change of the variable of integration of equation (7) gives

$$f(\theta) = r_o^2 \pi \sin\theta \left(\frac{\epsilon'}{\epsilon}\right)^2 \left(\frac{\epsilon}{\epsilon'} + \frac{\epsilon'}{\epsilon} - \sin^2\theta\right) \quad (9)$$

which is equation (7) integrated over all values of ϕ.

A computer program was written for equation (9) utilizing Simpson's rule to evaluate the definite integral:

$$\int_{\theta_a}^{\pi} f(\theta)d\theta \qquad \text{for values of } \theta_a = \pi/2,\ 2\pi/3 \text{ and } 3\pi/4 \tag{10}$$

This integral was evaluated for energies ranging from 10 Kev to 100 Mev to obtain quantitative results on the number of photons that would be scattered between the stated values of θ over the energy range of 10 Kev to 100 Mev.

The curves in Figures 12 and 13 show the per cent of the interacting photons which undergo the Compton effect and scatter at angles of θ greater than $\pi/2$ and $3\pi/4$, respectively, for various energies. It may be observed from Figure 13 that for photons with an energy of 0.1 Mev, only 14.6% of those undergoing the Compton effect will be scattered more than $\theta = 3\pi/4$. At this energy about 7.8% of the interacting gamma radiation undergoes the Compton effect. Therefore, a Compton eliminator system with a shield around 10.7 steradians - equivalent to $\theta = 3\pi/4$ - will eliminate all but 14% of the 7.8% initially interacting via the Compton effect. Therefore, over 98% of the observed interactions would be photoelectric and useable for energy analysis. The per cent of the photons interacting by Compton effect compared to the total interacting may be calculated from the data presented in Figure 10.

As another example, consider a photon energy of 1.0 Mev. For $\theta = 3\pi/4$, a Compton eliminator system would eliminate all but approximately 7.4% of photons undergoing the Compton effect, thus bringing the effective Compton cross section for single interactions into the range of the photoelectric cross section. Since in a thick crystal the scattered photons are

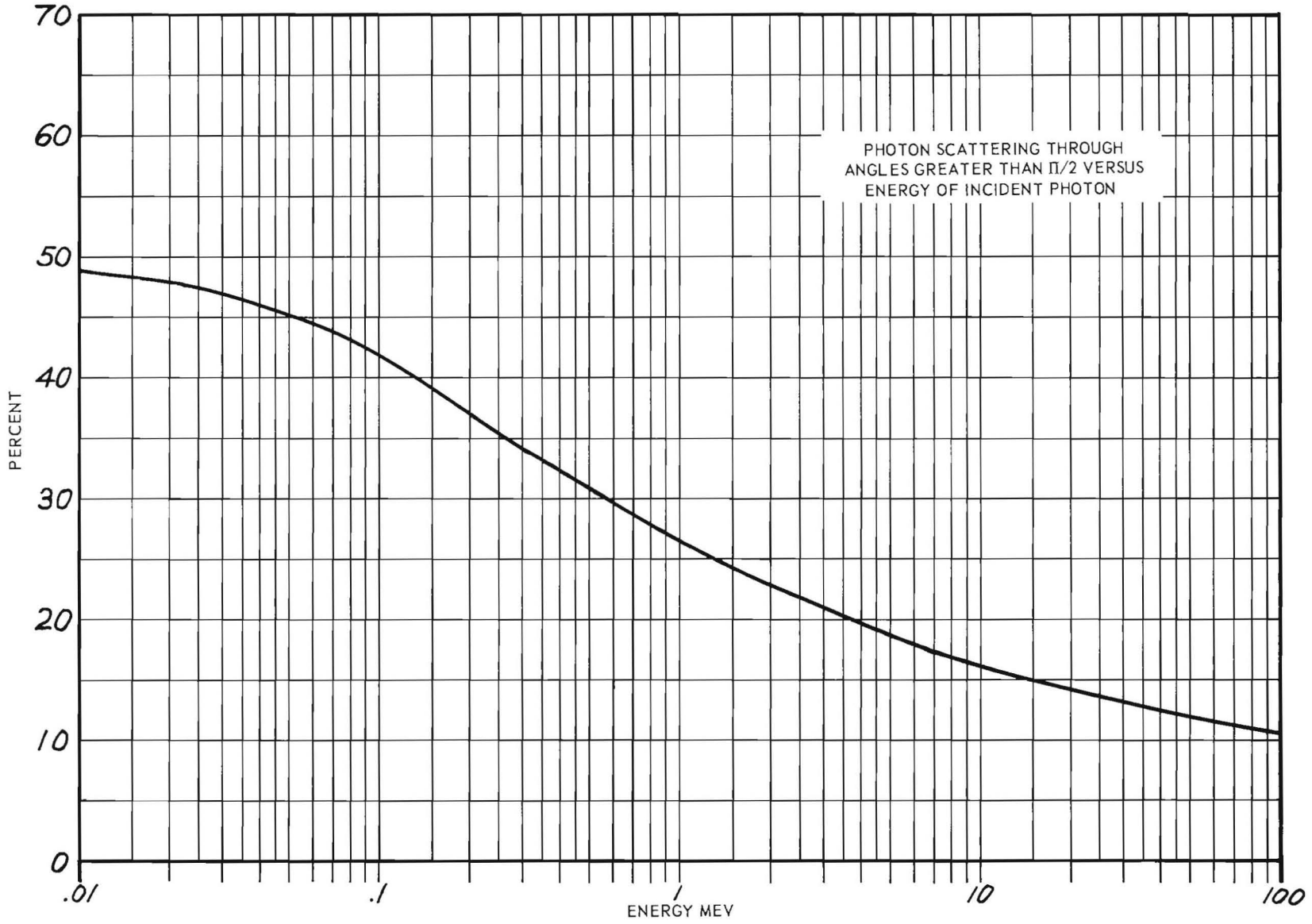


Figure 12. Photon Scattering through Angles Greater than $\pi/2$ Versus Energy of Incident Photon.

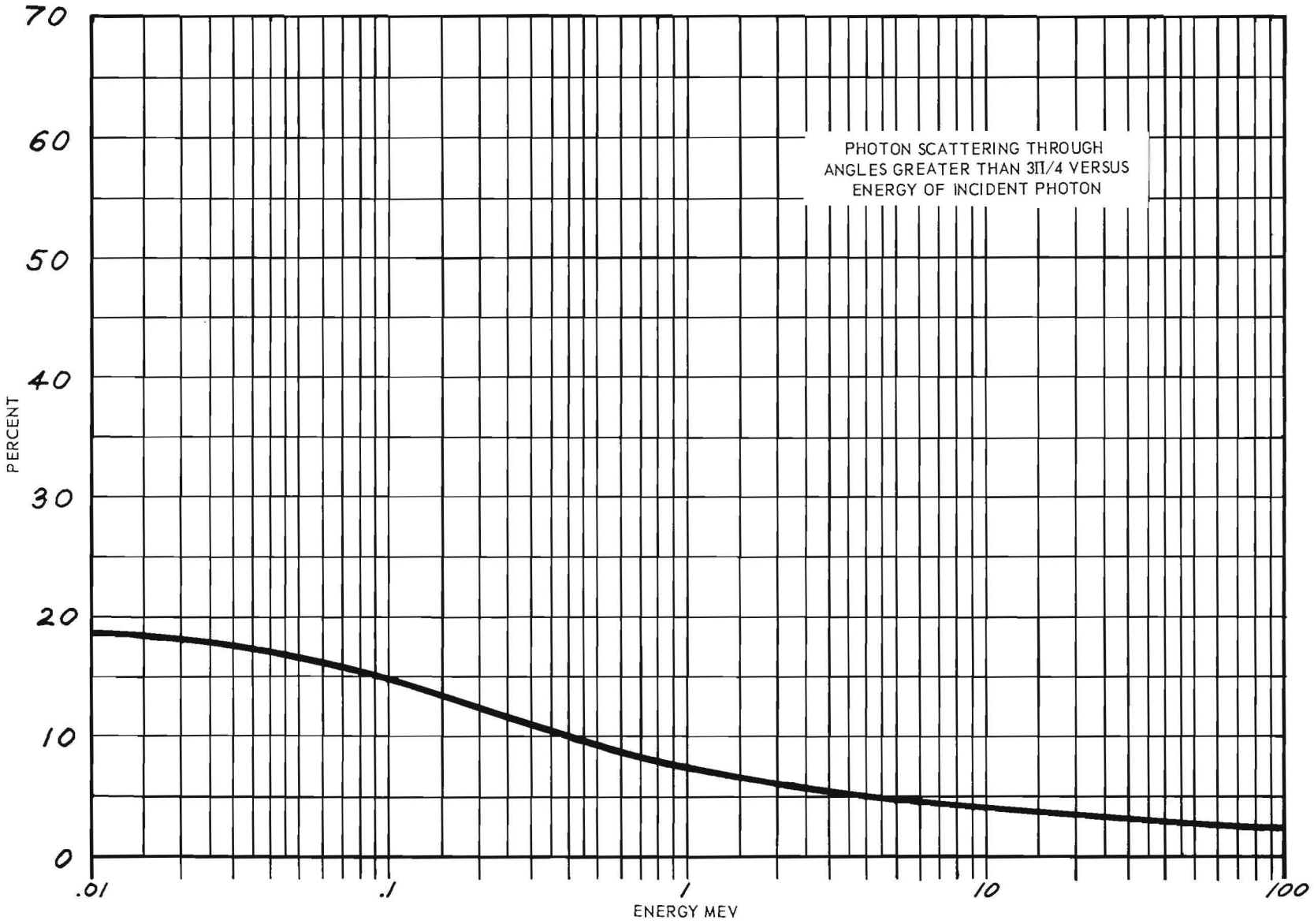


Figure 13. Photon Scattering Through Angles Greater than $3\pi/4$ Versus Energy of Incident Photon.

likely to undergo several interactions each of which decreases their energy and thus increases their absorption cross section, the overall probability for total energy absorption of an interacting gamma photon is much better than the preceding thin target analysis would reveal. Therefore, the Compton eliminator shield, in conjunction with a thick NaI crystal, would yield quite useful energy information in the range of interest.

D. Experimental Compton Eliminator Assembly

Studies utilizing the theory described above have been conducted using experimental apparatus constructed at this laboratory. A photograph of the assembly is shown in Figure 14. The initial experiments made with this apparatus have not produced the expected results.

It is postulated that the expected results were not produced because of the inadequate optical or detection efficiency of the shielding system. However the 5-in x 5-in plastic scintillator was the only large detector available, and time and funds did not permit the procurement of a scintillation crystal with the efficiency required for this application. A more elaborate system, including a NaI shield detector, is proposed for future experiments.

E. Neutron and Beta Monitoring Systems

The neutron and beta detection systems were described in a previous progress report (17). These systems were not completed due to increased effort directed toward completion of the gamma monitoring system in time for use in the fallout monitoring program. The neutron monitoring unit was completed but final testing could not be accomplished. The beta detection system progressed to the development stage, but little construction could be undertaken in the time allotted.

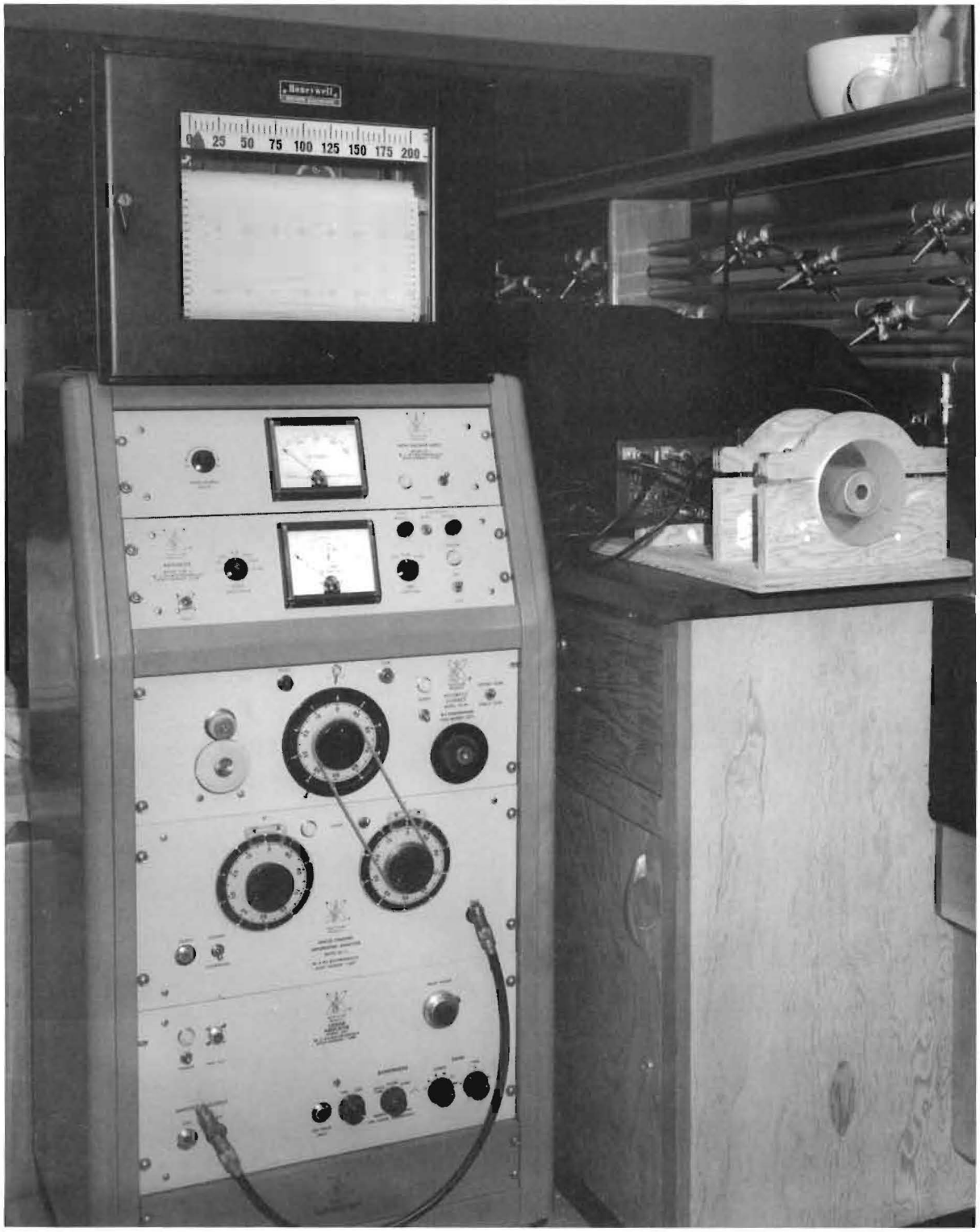


Figure 14. Experimental Compton Eliminator Assembly and Single Channel Gamma Analyzer.

IV. MEASUREMENT OF FALLOUT RADIATION

A. Scope of Services Performed

Careful and continuous studies have been conducted to measure the total flux density of the environmental background and to determine the activity of fallout particulates on the roof of the Civil Engineering Building at the Georgia Institute of Technology, Atlanta, Georgia. To assist the Division of Radiological Health's National Surveillance Network Program and to respond to the public health scientific needs created by the fallout from the Soviet weapons tests in the fall of 1961, the existing research program[6] on the development of low-level background detection underwent a change of direction to emphasize and continue work on radiation measurement activities. The change of direction in October 1961 resulted in the following major objectives:

(1) The Measurement of Background Gamma Flux Density

Qualitative determination of the gamma background is achieved by the measurement of the flux in three energy ranges.

(2) The Analysis of Alpha and Beta Activity of Fallout Particulates

At the time that the decision was made to participate in the Surveillance Network Program the three channel analyzer had just been completed and was undergoing trial runs. There was no time to determine the efficiency of the gamma detection system for photons of various energies. The equipment was initially operated manually, but during December 1961 the problems associated with automatic recording of the three channel spectrum were resolved successfully. Two air filter systems were obtained to carry out

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6 Research Contract SAph 76183 between the Division of Radiological Health, U.S.P.H.S., Washington 25, D.C. and the Georgia Institute of Technology, Atlanta 13, Georgia.

measurements of fallout particulates. One of these was an automatic filter tape air sampler[7], which employs a continuous membrane type filter tape from which the one inch samples were prepared. The other air sampling device is a Gast rotary type vacuum pump combined with a Millipore filter head which accommodates two inch filter samples. The NaI scintillation probe of the three channel gamma analyzer and the two air filter systems were located on the roof, while the requisite instrumentation was located in the nuclear counting laboratory on the top floor of the same building. Continuous read-out of the three channel gamma analyzer has already been mentioned. On March 1, 1962, the tape sampler was modified with a G.M.tube as shown in Figure 15, in order that each sample could be counted automatically while still on the tape, and if necessary, could be removed immediately for further laboratory analysis. The modification of the tape sampler included the mounting of an end window G.M. tube 2 mm from the filter tape. An automatic sequencer unit was designed to operate the tape sampler in conjunction with a scaler[8] and a printing timer[9]. Air was forced through the filter tape by a vacuum pump for a predetermined time interval. The sample was then moved under the G.M. tube and counted for a predetermined number of counts. It was planned to perform a correlation to determine the ratio of efficiencies between counting the samples in this manner and counting in a micromil gas flow G. M. counter[10]. This automatic counting method of the 1-in air filter sample

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7 Model M Hi-Flow Tape Sampler, manufactured by the Research Appliance Company, Allison Park, Pennsylvania.

8 Model 186 P, manufactured by the Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

9 Model C-111, Printing Timer, manufactured by Clary Corp., San Gabriel, California.

10 Model C-110B, manufactured by the Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

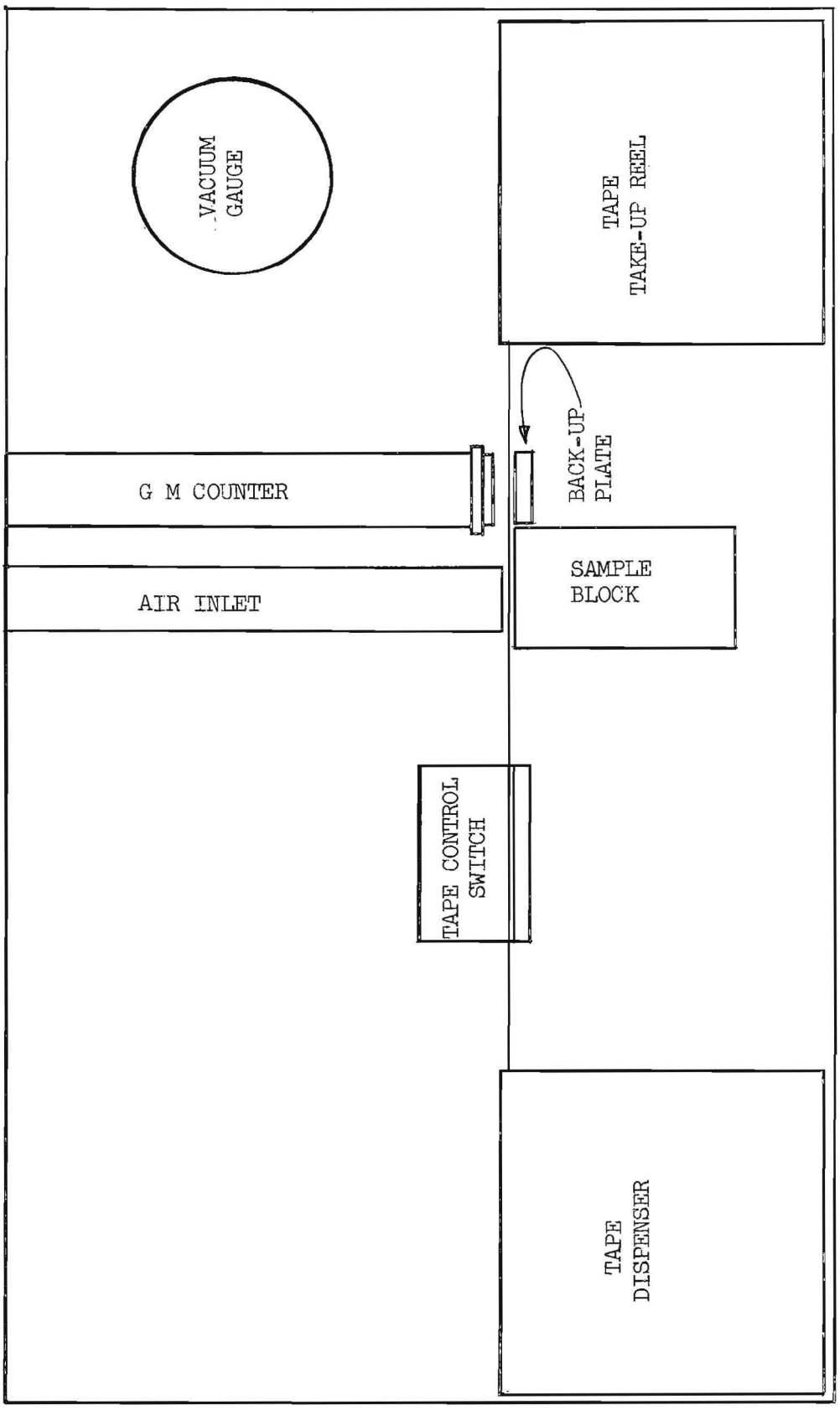

Figure 15. Automatic Air Sampling Unit

eliminated sample preparation time, except when an exceptionally active sample was found which could be subjected to delayed laboratory analysis. However, a continuous maintenance problem was created as the result of dust leaving the tape and contaminating the G.M. tube. At the end of the project, a mechanical modification was under way to alleviate this problem. It was planned to hold the tape down to a backing plate with a slight vacuum applied through the backing plate under the G.M. tube.

B. Details of Instrumentation Employed

(1) Total Gamma Dose by the Three Channel Analyzer

To facilitate the measurement of man's total immersion dose of gamma radiation in the environment, a transistorized three channel gamma analyzer has been designed, utilizing a 2-in by 2-in NaI scintillator[11]. This particular analyzer was developed at this laboratory and employs real time pulse amplitude analysis as compared to pulse height to width conversion techniques employed in the slower, multi-channel analyzers. This unit quantizes the background radiation into three energy levels, e.g., 0.1 to 0.5 Mev.; 0.5 to 1.0 Mev.; over 1.0 Mev. Photographs of the gamma analysis unit are shown in Figures 8 and 9. Changes in the counting rate of the high energy channels indicate a change in the cosmic ray background, whereas changes in the counting rate of the lower channels, if unaccompanied by changes in the high channel, indicate a change in the radiation flux derived from fallout. Since the detector efficiency is energy dependent, this energy quantization allows a much improved measure of ionizing background flux as compared to other means of survey measurement (e.g., geiger counters, etc.).

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[11] NaI crystal, type 7F8, manufactured by Harshaw Chemical Co., 1945 E. 47th Street, Cleveland Ohio.

When completely developed, this unit will be capable of unattended monitoring at remote locations. It will trigger an alarm at a preset level to warn of dangerous radiation levels and through energy discriminations the unit will provide some indication of the source of the radiation.

The information derived from the pulse analysis system was time-multiplexed onto a strip-chart recorder for subsequent processing.

(2) Particulates from One Inch Millipore Filter Samples

The nominal 1-inch filter samples (true diameter 1.6 cm) were initially collected over a one, two or three hour period by the Hi-Flo Tape Sampler using continuous Millipore tape, type WS with a pore size of 3.0 microns. After December 1, 1961, the sampling period was three hours. The tape was removed from the sampler every three or four days and the individual samples were glued to 1-inch diameter aluminum planchets. The samples were then counted for beta emission in a gas flow counter using Q gas[12]. The counting equipment consisted of two scalers[13], two automatic sample changers[14], and two printing timers[15]. Counting time for each sample ranged from 15 to 300 minutes. A photograph of the counting laboratory and some of the equipment employed in this work is shown in Figure 16.

(3) Particulates from Two Inch Millipore Filter Samples

The nominal 2-inch samples (actual diameter 3.46 cm) were collected over a one, two or three day period on Millipore type HA filter discs with

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12 Model D-47 Gas Flow Counter, manufactured by Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

13 Model 186 P Scaler manufactured by Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

14 Model C-111 and C-110 Time Interval Printers, manufactured by Clary Corp., San Gabriel, California.

15 Model C-110-B Sample Changer, manufactured by Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

Figure 16. Nuclear Counting Laboratory.

a pore size of 0.45 micron (the early samples were taken on type AA, 0.80 micron filter discs). The filter apparatus is a Gast Rotary Air Pump with a Millipore, Aerosol Open Type, filter holder. The dust samples were glued to 2-inch stainless steel planchets and counted in a Proportional Counter Converter[16] using P-10 gas for alpha-beta discrimination. Two Model 183 scalers[17] were used in conjunction with the proportional counter. Each sample was counted for at least 15 to 20 minutes.

C. Presentation of Data

In the tables and graphs presented in two interim reports (22)(23), the data collected from October, 1961 to July, 1962 has been compiled to show the counting rate in terms of counts per minute, as well as the activity expressed as μμc/cubic meter of air. The activities have been corrected for efficiency, which was found to be about 20% for the one inch nominal size samples and about 30% for the two inch nominal size samples.

1. Efficiency Calculations for 1-inch and 2-inch Diameter Air Samples

Counting efficiencies of both one and two inch samples were calculated by counting five selected air filter samples of each type, dosed with a known amount of Cs^{137}. Previously collected filter samples of low activity were used to approximate geometry and self-absorption as closely as possible.

Samples were prepared by:

(i) Recounting previously low activity samples to establish

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16 Model PCC-11A Proportional Counter Converter, a product of Nuclear Measurements Corp., 2460 N. Arlington Avenue, Indianapolis, Indiana.

17 Model 183 Scaler, manufactured by the Nuclear Chicago Corp., 333 East Howard Ave., Des Plaines, Illinois.

background (equipment background and sample activity),

(ii) Adding one drop of ethyl alcohol on the filter sample to assist distribution of the Cs^{137} solution over the dust sample, and

(iii) Adding an exact amount of Cs^{137} (450 μμc).

Samples were then counted four times; rotated 90° in the chamber after each count.

2. Sample Calculations for 1-inch and 2-inch Diameter Air Samples

Net cpm:

actual cpm - background cpm = net cpm

Flow rate:

$$0.50\ ft^3/min \times 60\ min/hr \times 0.02832\ m^3/ft^3 = 0.85\ m^3/hr \text{ of sampling}$$

cpm/m^3:

$$\frac{\text{net cpm}}{0.85\ m^3/hr \times \text{no. hrs. sampled}} = cpm/m^3$$

$\mu\mu c/m^3$:

$$cpm/m^3 \times \frac{1\ \mu\mu c}{2.22\ dpm} \times 1/\text{efficiency} = \mu\mu c/m^3$$

Efficiency values were found to be 18.4% and 29.6% for the one and two inch samples, respectively.

3. Variations Exhibited by Data

(i) One inch samples (3.0 μ)

A total of 1349 one inch air samples were collected. The results of their radiometric analyses are presented elsewhere (22) (23). Sample collection period and date, time elapsed before counting and the net beta counting rate for each sample is shown. In general, no alpha activity determinations were attempted for the one inch samples. All values expressed in $\mu\mu c/m^3$ have been corrected for efficiency, however, no correction has been supplied for decay.

Of the 1349 one inch samples, almost 10 per cent (125 samples) had activities exceeding 25 $\mu\mu c/m^3$. The majority of these (114) were in the range of 25 to 50 $\mu\mu c/m^3$. Eight samples were in the range of 51 to 100 $\mu\mu c/m^3$, and three samples had activities greater than 100 $\mu\mu c/m^3$. The grouping of these samples in ranges of activity is shown in Table I. The average monthly values are shown in Table II.

(ii) Two inch samples (0.45 μ):

A total of 172 two inch air samples were collected. The results of their radiometric analyses are presented elsewhere (22)(23). By means of alpha-beta discrimination, both alpha and beta activities could be obtained simultaneously. All other operating conditions and data analysis procedures were performed in identical manner to the one inch samples previously described.

Of the 172 two inch samples, 13 had an activity greater than 35 $\mu\mu c/m^3$. No samples were found having an activity greater than 56 $\mu\mu c/m^3$. Of these 13 samples, three were in the range from 50 to 55 $\mu\mu c/m^3$. The grouping of these samples in ranges of activity is shown in Table III. The average monthly values are shown in Table IV. Little direct correlation was observed among the one and two inch samples of highest activity.

(iii) Three channel gamma analysis

The data from the three channel gamma analyzer, with a fourth channel giving total dose rate, have been previously presented in graphic form (22). In February, 1962, the scintillation probe was modified after some careful studies to compensate for the temperature changes to which the probe was exposed on the roof of the Civil Engineering Building.

TABLE I

GROUPING OF ACTIVITIES OF ONE INCH SAMPLES OF AIR PARTICULATES
(Collected from October, 1961 to July, 1962)

Activity ($\mu\mu c/m^3$)	No. of Samples
0 - 9	593
10 - 19	535
20 - 29	156
30 - 39	39
40 - 49	15
50 - 59	1
60 - 69	2
70 - 79	1
80 - 89	3
90 - 99	1
> 100	3
Total	1,349

Note: 100 $\mu\mu c/m^3$ = 1 x 10^{-10} $\mu c/cc$

TABLE II

AVERAGE MONTHLY ACTIVITIES OF ONE INCH SAMPLES OF AIR PARTICULATES

Month	No. of Samples	Average Activity ($\mu\mu c/m^3$)
10/61	61	18.5
11/61	260	10.7
12/61	244	11.2
1/62	243	14.2
2/62	67	16.6
3/62	64	17.7
4/62	29	22.9
5/62	108	17.3
6/62	186	9.5
7/62	87	10.1
Total	1,349	

TABLE III

GROUPING OF ACTIVITIES OF TWO INCH SAMPLES OF AIR PARTICULATES
(Collected from October, 1961 to July, 1962)

Activity ($\mu\mu c/m^3$)	No. of Samples
0 - 9	38
10 - 19	69
20 - 29	35
30 - 39	23
40 - 49	4
50 - 55	3
Total	172

TABLE IV

AVERAGE MONTHLY ACTIVITIES OF TWO INCH SAMPLES OF AIR PARTICULATES

Month	No. of Samples	Average Activity ($\mu\mu c/m^3$)
10/61	2	11.4
11/61	17	27.7
12/61	23	16.4
1/62	15	12.7
2/62	16	19.7
3/62	22	20.9
4/62	25	24.9
5/62	23	19.0
6/62	21	12.5
7/62	8	14.4
	Total 172	

The crystal calibration was carried out by placing a Cs^{137} source 80 inches from the crystal and comparing the observed count rate to the actual gamma flux. The flux was determined by dividing the cross-sectional area of the sodium iodide crystal by the square of the distance from the source to the crystal and multiplying the result by the activity divided by 4π. Calibration by this method yielded an efficiency of 85% from a gamma flux of 0.662 Mev photons.

V. SUMMARY

1. Neutrons, electrons, and gamma radiation are three detectable products of cosmic radiation.

2. Variations of the above three products may prove a reliable index to cosmic ray activity.

3. Electrons (beta particles) and gamma rays are also produced from terrestrial sources, as are alpha particles.

4. A three channel gamma spectrum analyzer along with the necessary auxiliary equipment has been designed and developed.

5. A study of the problem of calibrating a NaI scintillator was made and experiments conducted to determine a method of constructing a detector sensitive only to those gamma photons totally absorbed by the NaI scintillator.

6. The gamma analysis unit was operated full time to make measurements of gamma radiation from fallout during the period October 1961 to July 1962.

VI. CONCLUSIONS

1. Four types of radioactive emanations are: alpha, beta, gamma rays and neutrons. They are produced by terrestrial and cosmic sources.

2. Beta particles are the predominant component of cosmic radiation, while alpha particles predominate from terrestrial radiation due to naturally occurring radioactivity.

3. By observing the variations of alpha, beta, gamma and neutron radiation levels and comparing their respective ratios, predictions as to the source of fluctuations might be made.

4. The gamma analyzer operates as designed and provides a reliable instrument for quantizing the gamma photons into three apparent energy bands.

5. Some problems are associated with calibrating a sodium iodide scintillator versus energy. For an absolute calibration, it is necessary to know the exact relationship between a given output pulse and the photon causing it. A feasible solution to this problem has been found and is described in this report.

VII. RECOMMENDATIONS

1. The gamma analysis unit is operational. It can be mass produced on short notice when an inexpensive gamma background analysis unit is needed.

2. The work on the neutron and beta systems should be continued so that a complete background monitor would be available.

3. There exists a need for further study of the problem of absolute energy calibration of a sodium iodide crystal.

4. A study of the directional characteristics of the cosmic ray flux is needed to accurately predict the efficiency of a specific detector geometry in a particular orientation when the crystal is to be used in measuring the background radiation flux.

5. The present state of knowledge in the field of alpha particle detection and dose evaluation is sufficiently advanced in comparison with the measurement of other types of ionizing radiation. Thus, it is recommended that the major investigative effort be directed toward the analysis of the other components of ionizing radiation.

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